February 22, 2000

Mr. James Knubel
Chief Nuclear Officer
Power Authority of the State of
New York
123 Main Street
White Plains, NY 10601

SUBJECT: RELIEF REQUEST NO. 17 - REQUEST FOR RELIEF FROM THE REQUIREMENTS OF 10 CFR 50.55a(g)(6)(ii)(A)(*2*) FOR AUGMENTED INSPECTION OF THE CIRCUMFERENTIAL WELDS IN THE REACTOR VESSEL OF THE JAMES A. FITZPATRICK NUCLEAR POWER PLANT (TAC NO. MA6215)

Dear Mr. Knubel:

By letter dated August 5, 1999, as supplemented November 24, 1999, you submitted Relief Request No. 17, requesting relief from performing the augmented inspections of circumferential welds in the reactor pressure vessel (RPV) of the James A. FitzPatrick Nuclear Power Plant. These inspections are required pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(*2*). In Attachment 1 to your letter of August 5, 1999, you indicated that 10 CFR 50.55a(g)(6)(ii)(A)(*5*) requires licensees to submit alternative programs to these augmented inspection requirements if it is determined that they cannot completely comply with the augmented inspection requirements of 10 CFR 50.55a(g)(6)(ii)(A)(*2*). You therefore submitted your relief request under the provisions of 10 CFR 50.55a(a)(3)(i), which allow licensees to propose alternative programs to the requirements of 10 CFR 50.55a if the program can be shown to provide an acceptable level of quality and safety in lieu of complying with the applicable inservice inspection requirements cited in the rule. Such alternative programs are approved by the Director of the Office of Nuclear Reactor Regulation, or his designee, on a case-by-case basis. In Attachment 1 to your letter of August 5, 1999, you indicated that this alternative program was based on the probabilistic fracture mechanics methods of BWR Vessel and Internals Project (BWRVIP) Topical Report No. TR-105697, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05). On November 24, 1999, you submitted additional information to clarify the contents of your evaluation.

The staff has determined that your alternative proposal for permanent deferment of the augmented inspections of the circumferential RPV shell welds is consistent with the guidance of Generic Letter 98-05, which provides the staff's guidance regarding submittal of BWRVIP-05 evaluations. The staff has also determined that your alternative proposal meets the probabilistic fracture mechanics acceptance criteria for deferring the inspections of the circumferential welds, which were summarized in the staff's safety evaluation on BWRVIP-05, dated July 30, 1999. The Nuclear Regulatory Commission staff therefore concludes that in regard to Relief Request No. 17, your probabilistic fracture mechanics analysis for the circumferential RPV welds provides an acceptable level of quality and safety in lieu of performing the required augmented volumetric inspections on the circumferential RPV welds. Therefore, the proposed alternatives may be authorized pursuant to 10 CFR 50.55a(a)(3)(i). Pursuant to 10 CFR 50.55a(a)(3)(i), the relief request is granted. The relief granted is authorized by law and will not endanger life or property, or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee if compliance with the actual inservice inspection requirements were imposed on the facility. The staff's evaluation and conclusions are contained in the enclosed safety evaluation.

Sincerely,

/RA by Peter Tam for/

Marsha K. Gamberoni, Acting Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-333

Enclosure: Safety Evaluation

cc w/encl: See next page

performing the required augmented volumetric inspections on the circumferential RPV welds. Therefore, the proposed alternatives may be authorized pursuant to 10 CFR 50.55a(a)(3)(i). Pursuant to 10 CFR 50.55a(a)(3)(i), the relief request is granted. The relief granted is authorized by law and will not endanger life or property, or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee if compliance with the actual inservice inspection requirements were imposed on the facility. The staff's evaluation and conclusions are contained in the enclosed safety evaluation.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ALTERNATIVES FOR EXAMINATION OF REACTOR PRESSURE VESSEL SHELL WELDS

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-333

1.0 INTRODUCTION

By letter dated August 5, 1999, (Relief Request No. 17) supplemented by a letter dated November 24, 1999, the Power Authority of the State of New York (PASNY), licensee for the James A. FitzPatrick Nuclear Power Plant, requested that the Nuclear Regulatory Commission (NRC) approve an alternative to performing circumferential shell weld examinations on the reactor pressure vessel (RPV) welds (Ref. 1). These examinations are required by Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), and by the augmented examination requirements of Section 50.55a(g)(6)(ii)(A)(2) to Title 10 of the Code of Federal Regulations (10 CFR 50.55a(g)(6)(ii)(A)(2)). The alternative was proposed pursuant to the provisions of 10 CFR 50.55a(a)(3)(i) and is consistent with the guidance provided in Generic Letter 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief From Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," dated November 10, 1998 (Ref. 2), and the staff's evaluation of the BWRVIP-05 report issued July 28, 1998 (Ref. 3).

1.1 Regulatory Requirements

Pursuant to the requirements of 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components are to meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of the ASME Code, Section XI, incorporated by reference in 10 CFR 50.55a(b) on the date 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein.

It is required by 10 CFR 50.55a(g)(6)(ii)(A) that licensees perform an augmented RPV shell weld examination as specified in the 1989 Edition of Section XI of the ASME Code. The final Rule was published in the *Federal Register* on August 6, 1992 (57 FR 34666). By incorporating into the regulations the 1989 Edition of the ASME Code, the NRC staff required that licensees perform volumetric examinations of "essentially 100 percent" of the RPV pressure-retaining shell weld, during all inspection intervals. It is stated in 10 CFR 50.55a(a)(3) that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

1.2 BWRVIP-05 Report

By letter dated September 28, 1995, as modified and supplemented by letters dated June 24 and October 29, 1996, and May 16, June 4, June 13 and December 18, 1997, the Boiling Water Reactor Vessel and Internals Project (BWRVIP), submitted the proprietary report BWRVIP-05, "BWR Vessel and Internals Project, BWR Reactor Vessel Shell Weld Inspection Recommendations" (Ref. 4). As modified, the BWRVIP report proposed to reduce the scope of inspection of BWR RPV welds from essentially 100 percent of all RPV shell welds to examination of essentially 100 percent of the axial (i.e., longitudinal) welds and essentially zero percent of the circumferential RPV shell welds, except at the intersection of the axial and circumferential welds, thereby including approximately 2-3 percent of the circumferential welds. In addition, the report provided proposals to revise ASME Code requirements for successive and additional examinations of circumferential welds, provided in paragraph IWB-2420(b) of Section XI of the ASME Code.

On July 28, 1998, the NRC staff issued a safety evaluation of the BWRVIP-05 report (Ref. 3). This evaluation concluded that the failure frequency of RPV circumferential welds in boiling water reactors (BWRs) was sufficiently low to justify elimination of inservice inspection of these welds. In addition, the evaluation concluded that the BWRVIP proposals on successive and additional examinations of circumferential welds were acceptable. The evaluation indicated that examination of the circumferential welds shall be performed if axial weld examinations reveal an active, mechanistic mode of degradation.

In the BWRVIP-05 report (Ref. 4), the BWRVIP concluded that the conditional probabilities of failure for BWR RPV circumferential welds are orders of magnitude lower than that of the longitudinal welds. As a part of its review of the report, the NRC conducted an independent risk-informed, probabilistic fracture mechanics assessment of the results presented in the BWRVIP-05 report. The staff's assessment conservatively calculated the conditional probability of failure from RPV axial and circumferential welds during the (current) initial 40-year license period and at conditions approximating an 80-year vessel lifetime for a BWR nuclear plant, as indicated in Tables 2.6-4 and 2.6-5, respectively (Ref. 3). The failure frequency for a reactor pressure vessel is calculated as the product of the frequency for the critical (limiting) transient event and the conditional probability of failure for the weld.

The staff determined the conditional probability of failure for longitudinal and circumferential welds in BWR vessels fabricated by Chicago Bridge and Iron (CB&I), Combustion Engineering (CE), and Babcock and Wilcox (B&W). The analysis identified a cold over-pressure event in a foreign reactor as the limiting event for BWR RPVs, with the pressure and temperature from this event used in the probabilistic fracture mechanics calculations. The staff estimated that the probability for the occurrence of the limiting over-pressurization transient was 1×10^{-3} per reactor year. For each of the vessel fabricators, Table 2.6-4 of the staff's evaluation (Ref. 3) identifies the conditional failure probabilities for the plant-specific conditions with the highest projected reference temperature (for that fabricator) after the initial 40-year license period.

1.3 Generic Letter 98-05

On November 10, 1998, the NRC issued Generic Letter (GL) 98-05 "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief From Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds." GL 98-05 stated that BWR licensees may request permanent (i.e., for the remaining term of operation under the existing, initial, license) relief from the inservice inspection requirements of 10 CFR 50.55a(g) for the volumetric examination of circumferential reactor pressure vessel welds (ASME Code Section XI, Table IWB-2500-I, Examination Category B-A, Item 1.11, "Circumferential Shell Welds"), upon demonstrating that:

(1) at the expiration of the license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the NRC staff's July 28, 1998, safety evaluation, and

(2) licensees have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the NRC staff's July 28, 1998, safety evaluation.

Licensees would still need to perform the required inspections of "essentially 100 percent" of all axial welds.

2.0 **INFORMATION PROVIDED BY LICENSEE**

This section describes the Code requirements and the components for which the licensee is seeking relief, the basis for the relief request, and a demonstration by the licensee that the criteria for relief are satisfied.

2.1 Code Requirements for Which Relief Is Sought

The licensee identifies the following Code requirements from which relief is sought:

(1) ASME Section XI, 1989 Edition (no addenda), Table IWB-2500-1, Examination Category B-A, Item No. B1.11, volumetric examination of reactor pressure vessel circumferential welds. Permanent relief (i.e., for the remaining term of operation under the existing license) is requested.

2.1.1 Components for Which Relief Is Sought

The requested permanent relief from the Table IWB-2500-1 requirements applies to:

ISI Class 1, Code Category B-A, “Pressure Retaining Welds in Reactor Vessel,” Item B1.11, “Circumferential Shell Welds.”

RPV Circumferential Welds: VC-1-2, VC-2-3, VC-3-4, VC-4-BH-1.

2.2 Licensee's Basis for Relief

The licensee's request is based upon provisions in the NRC SE for the BWRVIP-05 Report (Ref. 3) and the guidance outlined in GL 98-05 (Ref. 2). These documents provide the basis for the elimination of inservice inspections of BWR RPV circumferential shell welds.

As described previously, GL 98-05 provides two criteria that relief request applicants must demonstrate, based upon the limiting conditional failure probability of the applicant's circumferential welds and implementation of operator training and established procedures to limit the frequency of cold over-pressure events. These criteria are intended to demonstrate that the conditions at the applicant's plant are bounded by those in the safety evaluation (Ref. 3).

The NRC SE for the BWRVIP-05 Report (Ref. 3) evaluated the conditional failure probability of circumferential welds for the limiting plant-specific case of BWR RPVs manufactured by different vendors, including Combustion Engineering (CE), using the highest mean irradiated RT_{NDT} to determine the limiting case.

Since the FitzPatrick RPV was fabricated by CE, the relief request compared the mean irradiated RT_{NDT} for FitzPatrick to that for the limiting CE case described in Table 2.6-4 of Ref. 3. As illustrated in Table 1, the mean RT_{NDT} for FitzPatrick is lower than that for the limiting CE case, and the licensee concluded that the conditional failure probability for the FitzPatrick circumferential welds is bounded by the conditional failure probabilities in the staff's safety evaluation report through the end of the current license period.

Table 1: Comparison of FitzPatrick Circumferential Weld and the USNRC Limiting Plant-Specific Analysis from Table 2.6-4 of Ref. 3
EFPY = Effective Full Power Years

PARAMETER	FITZPATRICK COMPARATIVE DATA AT 32 EFPY (BOUNDING CIRC WELD)	USNRC LIMITING PLANT-SPECIFIC ANALYSIS DATA AT 32 EFPY (Ref. 3) (CE "CEOG")
Fluence (10^{19} n/cm^2)	1.61	2.0
Initial RT_{NDT} (°F)	-50	0
Chemistry Factor (°F)	209.1	172.2
Cu (Wt. %)	0.337	0.183
Ni (Wt. %)	0.609	0.704
ΔRT_{NDT} (°F)	108.5	98.1
Mean RT_{NDT} (°F) [Initial RT_{NDT} + ΔRT_{NDT}]	58.5	98.1

To satisfy the second condition of GL 98-05 for relief consideration, the licensee reviewed the high pressure injection sources, administrative controls, and operator training regarding a cold over-pressure event. The licensee has assessed the systems that could lead to a cold over-pressurization of the FitzPatrick RPV. These include the high pressure core injection (HPCI), the reactor core isolation cooling (RCIC) system, the feedwater system, the control rod drive (CRD) system, and the standby liquid control (SLC) system.

The licensee stated that the feedwater system, the HPCI system and the RCIC system are all driven by steam turbine. During cold shutdown conditions, no steam is available to operate these systems, and therefore, they could not cause a cold over-pressure event.

The SLC system is an additional high pressure source. However, there are no automatic starts associated with the SLC system. The system is only initiated by manual operator action in accordance with the plant emergency operating procedures or during controlled test conditions, therefore, inadvertent manual initiation of SLC is an unlikely event. In addition, in the event of manual initiation during shutdown, the SLC injection rate of approximately 50 gpm would allow operators sufficient time to control reactor pressure.

During normal cold shutdown conditions, RPV level and pressure are controlled with the CRD system and the reactor water cleanup (RWCU) system using a “feed and bleed” process. The licensee has procedures in place to reduce the likelihood of a violation of the pressure-temperature limits, such as requiring the opening of the head vent valves after the reactor has been cooled to less than 212 °F. In addition, the slow injection rate of the CRD system under these conditions (<60 gpm), would allow the operator sufficient time to react to unanticipated changes in RPV level.

The CRD system and the RWCU system are also used to control RPV level and pressure during pressure testing of the RPV. The pressure increase is limited to 30 psig per minute which minimizes the likelihood of exceeding the pressure-temperature limits during the test.

In all cases, the operators are trained in methods of controlling water level within specified limits in addition to responding to abnormal water level conditions during shutdown. The licensee also stated that procedures and administrative controls for reactor temperature, level, and pressure are in place to minimize the potential for RPV cold over-pressure events. Plant-specific procedures have been established to provide guidance to the operators regarding compliance with the Technical Specification pressure-temperature limits.

2.3 <u>Licensee's Proposed Alternative Examination</u>

Pursuant to 10 CFR 50.55a(a)(3), the licensees are allowed to propose alternatives to the requirements of 10 CFR 50.55a(g). The licensee proposed, as an alterative, to perform vertical weld examinations and incidental examination of 2-to-3 percent of the intersecting circumferential shell welds to the maximum extent possible based on accessibility. The licensee would permanently defer examination of the circumferential welds until expiration of the plant’s current operating license.

3.0 EVALUATION

The staff's review focused on confirming that the licensee has adequately documented that the conditions for relief outlined in the SE to the BWRVIP-05 Report and GL 98-05 are satisfied.

3.1 Bases for Granting Relief From Complying with the Augmented Inspection Requirements for Circumferential RPV Welds

3.1.1 Circumferential Weld Conditional Failure Probability

The staff SE (Ref. 3) provides a limiting conditional failure probability of 6.34 x 10^{-5} per-reactor-year for a limiting plant-specific mean RT_{NDT} of 98.1 °F for CE-fabricated RPVs. Comparing the information in the NRC Reactor Vessel Integrity Database (RVID) with that submitted in the relief request, the staff has confirmed that the mean RT_{NDT} of the circumferential welds at FitzPatrick is projected to be 58.5 °F at the end of the current license. In this evaluation, the chemistry factor, ΔRT_{NDT}, and mean RT_{NDT} were calculated consistent with the guidelines of Regulatory Guide 1.99, Revision 2. The calculated value of mean RT_{NDT} for the circumferential welds at FitzPatrick is significantly lower than that for the limiting plant-specific case for CE-fabricated RPVs, indicating that the conditional failure probability of the FitzPatrick circumferential welds is much less than 6.34 x 10^{-5} per-reactor-year.

3.1.2 Cold Over-pressure Transient Probability

The staff concludes that a nondesign-basis cold over-pressure transient is unlikely to occur at FitzPatrick, and that the information provided by the licensee regarding the FitzPatrick high pressure injection systems, operator training, and plant-specific procedures provide a sufficient basis to support approval of the alternative examination request.

4.0 CONCLUSIONS

The staff has reviewed the licensee's submittal and finds that the licensee has provided an acceptable demonstration that the appropriate criteria in GL 98-05 and the staff's evaluation of the BWRVIP-05 report have been satisfied regarding permanent relief (i.e., for the remaining term of operation under the initial, existing license) from inservice inspection requirements for the volumetric examination of reactor pressure vessel circumferential welds, ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11.

The NRC staff concludes that authorization of the licensee's alternative examinations would provide assurance of structural integrity and, therefore, an acceptable level of quality and safety. Accordingly, pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5) and 10 CFR 50.55a(a)(3)(i), the licensee's proposed alternative examination is authorized. The refief granted is authorized by law and will not endanger life or property, or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee if compliance with the actual inservice inspection requirements were imposed on the facility.

Principal Contributor: M. Banic

Date: February 22, 2000

5.0 REFERENCES

1. Letter to U.S. NRC Document Control Desk, from J. Knubel, (PASNY), Subject: James A. FitzPatrick Nuclear Power Plant (Relief Request No. 17), Proposed Alternatives in Accordance with 10 CFR 50.55a(a)(3)(i) for Reactor Pressure Vessel Circumferential Shell Weld Examinations dated August 5, 1999.

2. Generic Letter 98-05, "Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief From Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," dated November 10, 1998.

3. "Evaluation by the Office of Nuclear Reactor Regulation Related to the Review of the Topical Report by the Boiling Water Reactor Vessel and Internals Project: BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," dated July 28, 1998.

4. EPRI Technical Report TR-105697, "Boiling Water Reactor Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)," dated September 28, 1995.